Report No. ACNP-63027

PATHFINDER ATOMIC POWER PLANT

TECHNICAL PROGRESS REPORT

APRIL 1963 – JUNE 1963

Submitted to

U. S. ATOMIC ENERGY COMMISSION

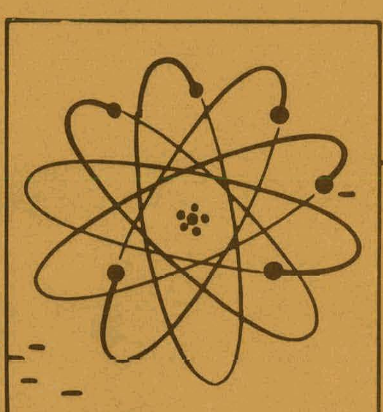
NORTHERN STATES POWER COMPANY

and

CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
Milwaukee 1, Wisconsin



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DISCLAIMER

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ALLIS-CHALMERS MANUFACTURING COMPANY

Under
Agreement dated 2nd Day of May 1957, as Amended
between
Allis-Chalmers Mfg. Co. & Northern States Power Co.
under
AEC Contract No. AT(11-1)-589

September 30, 1963

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FOREWORD

This is one of a series of reports covering technical progress on the research and development program being performed in connection with the design of the Pathfinder Atomic Power Plant. This plant will be located at a site near Sioux Falls, South Dakota and is scheduled for operation in 1963. Owners and operators of the plant will be the Northern States Power Company of Minneapolis, Minnesota.

The U. S. Atomic Energy Commission, through Contract No. AT(11-1)-589 with Northern States Power Company, and Central Utilities Atomic Power Associates* (CUAPA), are sponsors of the research and development program.

Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin, under contract with Northern States Power Company, is performing the research, development, and design; and will construct the plant including the reactor, which is designated the Controlled Recirculation Boiling Reactor (CRBR) with Nuclear Superheater. Pioneer Service and Engineering Company of Chicago, Illinois is providing the architect-engineer services to Allis-Chalmers. Portions of the R & D program, particularly in connection with fuel development have been subcontracted by Allis-Chalmers.

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* CUAPA MEMBER COMPANIES:

Interstate Power Company
Iowa Power and Light Company
Iowa Southern Utilities Company
Madison Gas and Electric Company
Northern States Power Company
Northwestern Public Service Co.
Otter-Tail Power Company
St. Joseph Light & Power Co.
Western Power and Gas Company
Wisconsin Public Service Corp.

DESIGN DATA

CRBR WITH NUCLEAR SUPERHEATER

Plant

Power, boiler region	157,200 kw
Power, superheater region	42,400 kw
Steam flow at rated power	616,125 lbs/hr
Total core power	199,600 kw
Gross electrical capability	66,000 kw
Net electrical output	62,000 kw
Net efficiency	31.0 per cent
Steam outlet pressure (reactor)	535 psig
Reactor operating pressure	600 psig
Temperature, boiler region	489 F
Outlet temperature, superheater region	825 F
Gross heat rate	10,199 Btu/kw-hr
Reactor building size	50 ft dia x 120 ft

Reactor

Vessel size (over-all)	11 ft 6 in o.d. 36 ft 1 in
Total core dimensions	6 ft x 6 ft
Dimensions of superheater region	6 ft x 30 in
Fuel, boiler (Zr-2 clad)	approx. 2.2 per cent enriched UO_2
Fuel, superheater (S.S. clad)	approx. 93 per cent enriched UO_2
Fuel, loading, boiler (U-235)	145.6 kg
Fuel, loading, superheater (U-235)	42 kg
Power density (boiler core coolant)	87 kw/liter
Average heat flux, boiler region	128,000 Btu/hr-ft^2
Average heat flux, superheater region	77,800 Btu/hr-ft^2
Maximum heat flux, boiler region	462,000 Btu/hr-ft^2
Maximum heat flux, superheater region	245,000 Btu/hr-ft^2
Recirculation rate	65,000 gpm
Recirculation pump power	823 kw
Neutron flux	approx. 5×10^{13} n/cm^{2}sec

3. NUCLEAR ANALYSIS

3.1 REACTOR PHYSICS (STATICS)

The objective of this project is to perform physics calculations such as computer programming and operation, and to determine the critical mass, neutron and gamma flux and power distribution, enrichment, coefficients of reactivity, control rod effectiveness, and conversion ratio with respect to a Pathfinder core with an integral high-enrichment superheater region. An additional objective is to determine shielding requirements for the Pathfinder plant.

3.1.1 PATHFINDER FIRST CORE HIGH ENRICHED (3.2 w/o U-235) BOILER FUEL ELEMENTS

In the last Pathfinder Quarterly Report (ACNP-63016) the use of boron stainless steel plates for negative reactivity shimming was described. In addition to these negative shims, thirty-two boiler fuel elements with a higher enrichment (3.2 w/o U-235) are available for positive reactivity insertion in the first Pathfinder core. To prevent an increase in the maximum boiler heat flux at full power, these thirty-two elements are limited to insertion in those core locations not adjacent to a control rod channel. Figure 3.1 shows the available core locations.

A previous analysis (described in Pathfinder Quarterly Progress Report ACNP-6117) determined the reactivity worth of all thirty-two elements when inserted in the reference core for hot operating condition. For that analysis, two-dimensional two-group diffusion theory calculations were utilized. The effect on radial power distribution was also analyzed.

A nuclear analysis was completed during the past Quarter (April-June, 1963) to determine the following additional information:

1. Reactivity worth for all thirty-two elements in the cold critical core...

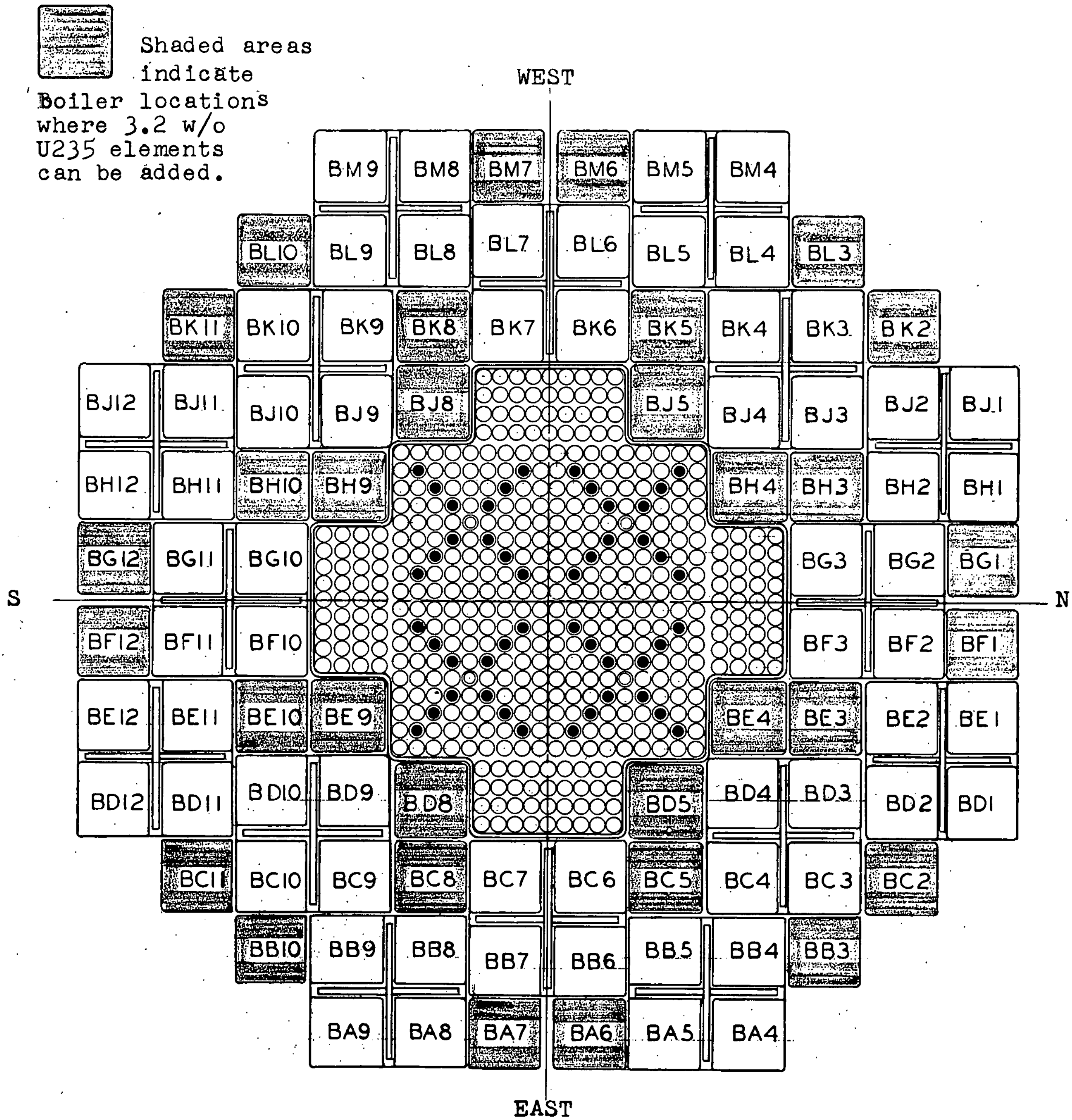


Fig. 3.1...Boiler Element Identification

2. Relative worth of individual boiler element locations for the cold and hot operating conditions...

3. The effect of the addition of these higher enriched fuel elements on the boiler and superheater power distribution.

Perturbation theory was used to determine the relative worth of the boiler fuel element locations for both the cold and hot operating conditions. Two-dimensional (XY), two-group adjoints and fluxes had been obtained for analysis of the worth of the boron stainless steel shims as described in ACNP-6117. They were obtained for the cold all-control-rods-in core, and the hot voided all-control-rods-out core. Using these adjoints and fluxes for the same core condition, and the following expression:

$$\Delta k/k = \frac{1}{F}\Big[-\int \phi_2{}^*\phi_2\,\delta\Sigma_{a_2}\,dV + \int \phi_1{}^*\phi_2\,\delta\nu\Sigma_{f_2}\,dV$$

$$+ \int \phi_1{}^*\phi_1\,\delta\nu\Sigma_{f_1}\,dV - \int \phi_1{}^*\phi_1\,\delta\Sigma_{a_1}\,dV + \int \phi_1(\phi_2{}^* - \phi_1{}^*)\,\delta\Sigma_{R1}\,dV\Big]$$

$$F = \int \phi_1{}^*\phi_2\,\nu\Sigma_{f2}\,dV + \int \phi_1{}^*\phi_1\,\nu\Sigma_{f1}\,dV$$

the relative worth of the fuel element locations in the boiler were determined and are shown in Table 3.1. The core locations in the table represent all of the available boiler element locations in each core octant. The 3.2 w/o elements will be loaded symmetrically by octant into the core. The maximum worth location is normalized to 1.0 for each condition.

Table 3.1 - RELATIVE WORTH OF BOILER ELEMENT LOCATIONS

Boiler Element* Location in Core	Relative Worth Cold...All Rods In	Relative Worth, Hot, Voided...All Rods Out
BA 7**	.25	.44
BA 8	.13	.40
BA 9	.06	.28
BB 7	.42	.83
BB 8	.50	.77
BB 9	.32	.61
BB 10**	.22	.39
BC 7	.63	.95
BC 8**	1.00 (Max. Worth)	.87
BC 9	.66	.93
BC 10	.38	.69
BD 8**	.89	.81
BD 9	.84	1.00 (Max. Worth)

* For identification of core locations, see Fig. 3.1
** Possible 3.2 w/o element locations

The reactivity worth had been obtained previously with all of these elements added to the reference core in their specified locations (Fig. 3.1) for the hot operating condition with all control rods withdrawn. During reactor operation, the excess reactivity will be controlled by insertion of groups of the outer eight control rods. Therefore, the reactivity worth and, in particular, the effect that these elements have on the radial distribution has now been analyzed with the eight outer control rods inserted.

This analysis was made using explicit PDQ-XY calculations, and the results are shown in Table 3.2. The worth of these elements rose from +.023 $\Delta k/k$ to +.029 $\Delta k/k$ with the insertion of the control rods, but no adverse effect on power peaking was noted. With or without the control rods inserted, a slight increase in the superheater power peak-to-average -- and a slight decrease in the boiler power peak-to-average -- is obtained when these elements are added.

Table 3.2 - PATHFINDER PERFORMANCE WITH 3.2 W/O U-235 ELEMENTS ADDED

Core Description	k_{eff}	Δk_{eff}	Superheater Power Factor	Superheater P/A Power	Boiler P/A Power
Reference Core, No Control Rods Inserted	1.060	-----	.1395	1.169	1.737
32 3.2 w/o Elements added, No control rods inserted	1.083	.023	.1340	1.174	1.636
Reference Core 8 Outer Rods Inserted	1.004	----	.1698	1.371	2.240
32 3.2 w/o Elements added, 8 Outer Control Rods Inserted	1.033	.029	.1542	1.394	2.138

Using perturbation theory normalized to successive two-dimensional calculation results, the reactivity worth of the 3.2 w/o enriched elements inserted in the cold all-rods-in core has been approximated. In order to maintain core symmetry, these elements will be utilized in groups of eight. The worth of each of the four groups has been approximated as mentioned above for the cold and hot operating condition - as shown in Table 3.3. If positive shim is necessary for full power operation, it is desirable to insert 3.2 w/o fuel elements into those locations that yield the least reactivity gain for the cold shutdown core and the most reactivity gain for the hot operating core with all control rods withdrawn -- the end of life control rod condition.

Table 3.3 - EXPECTED REACTIVITY WORTH OF GROUPS OF HIGH ENRICHED ELEMENTS

Core Locations Where Elements are Added (Fig. 3.1)	Δk/k Cold Core Condition	Δk/k Hot Operating All Rods Out
BA 7, BA 6, BF 1, BF 12 BG 1, BG 12, BM 6, BM 7	+ .0035	+ .0040
BB 3, BB 10, BC 2, BC 11 BK 2, BK 11, BL 3, BL 10	+ .0031	+ .0036
BC 5, BC 8, BE 3, BE 10 BH 3, BH 10, BK 5, BK 8	+ .0140	+ .0080
BD 5, BD 8, BE 4, BE 9 BH 4, BH 9, BJ 5, BJ 8	+ .0124	+ .0074

Based on the results in Table 3.3, 3.2 elements will be inserted just in locations BA 7, etc. ....then locations BB 3, etc...then locations BD 5, etc... and finally locations BC 5, etc.

PART B

POST CONSTRUCTION R&D

1.0 INITIAL STABILITY AND PERFORMANCE TESTS ...POST CONSTRUCTION R&D

The objectives of this project are to design and fabricate a special oscillator rod and drive mechanism together with suitable instrumentation and recording equipment to measure and record the resulting variations in neutron flux, and to conduct oscillator tests to verify dynamic performance calculations and to determine experimentally the stability of the reactor system. In addition, certain measurements of power, flows, pressures, and temperatures will be made to determine the transient response of the reactor system. The data will be analyzed to determine certain dynamic parameters.

1.0.1 DYNAMICS

1.0.1.1 In-Core Ion Chambers

The in-core ion chamber system which has been previously reviewed (ACNP-62005, Pathfinder Progress Report, October-December 1961) was brought to about 80 per cent completion during the present quarter. For review, Figure 1.1 shows the overall layout of the mechanical features of the system ...Figure 1.2 shows the instrumentation system.

The mechanical support structure has been successfully placed in the reactor, with the exception of the dummy fuel elements. These dummy superheater elements are being fabricated and are being wrapped with a thin silver foil. This foil will be sandwiched between thin stainless steel cylinders and fabricated to the same end-welding specifications as the superheater fuel elements in order to prevent silver contamination of the reactor system. The silver is necessary to provide neutron absorption, which will thus prevent undesirable flux peaking in adjacent superheater elements.

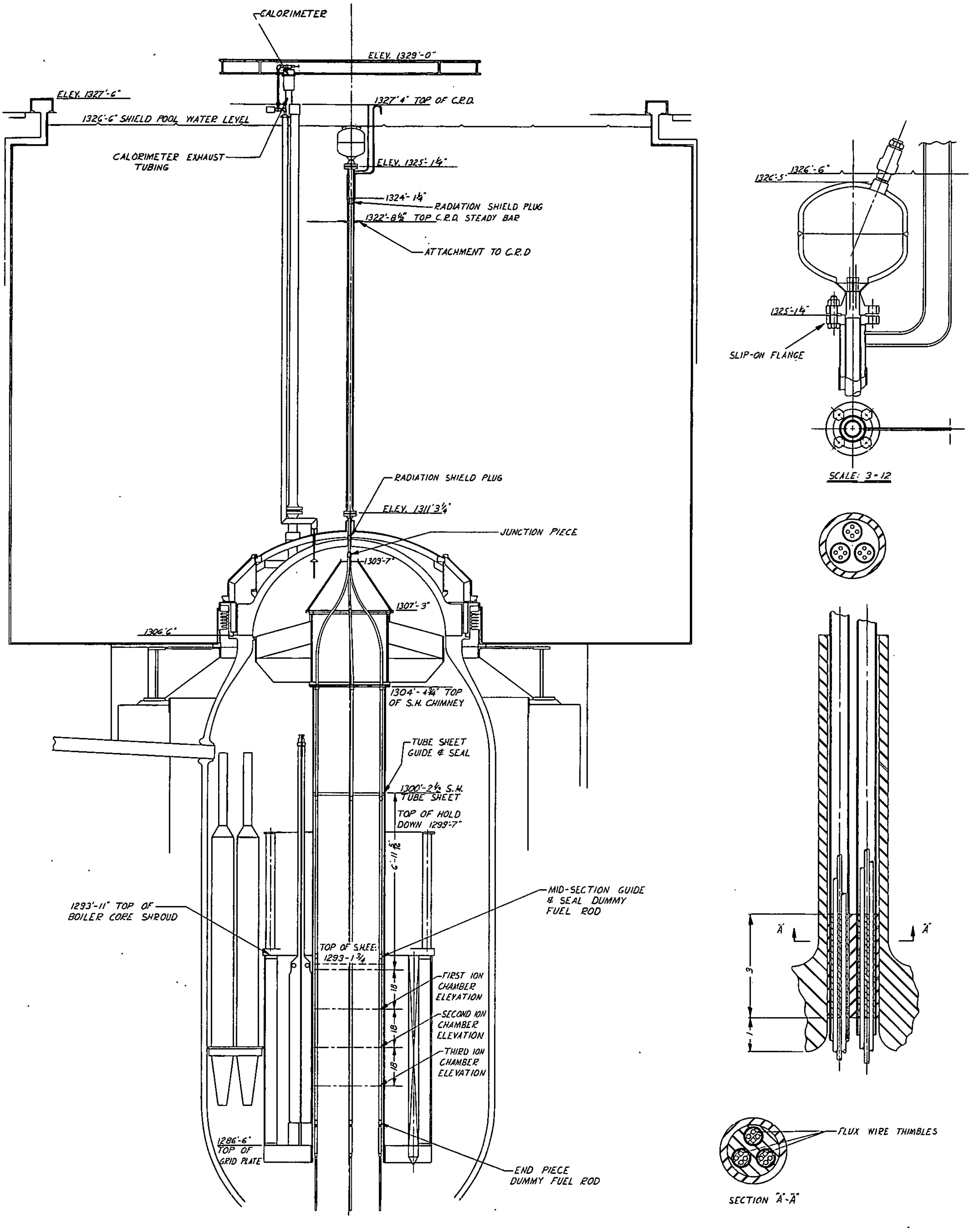


Fig. 1.1...Ion Chamber Support and Guide Structure (43-002-464)

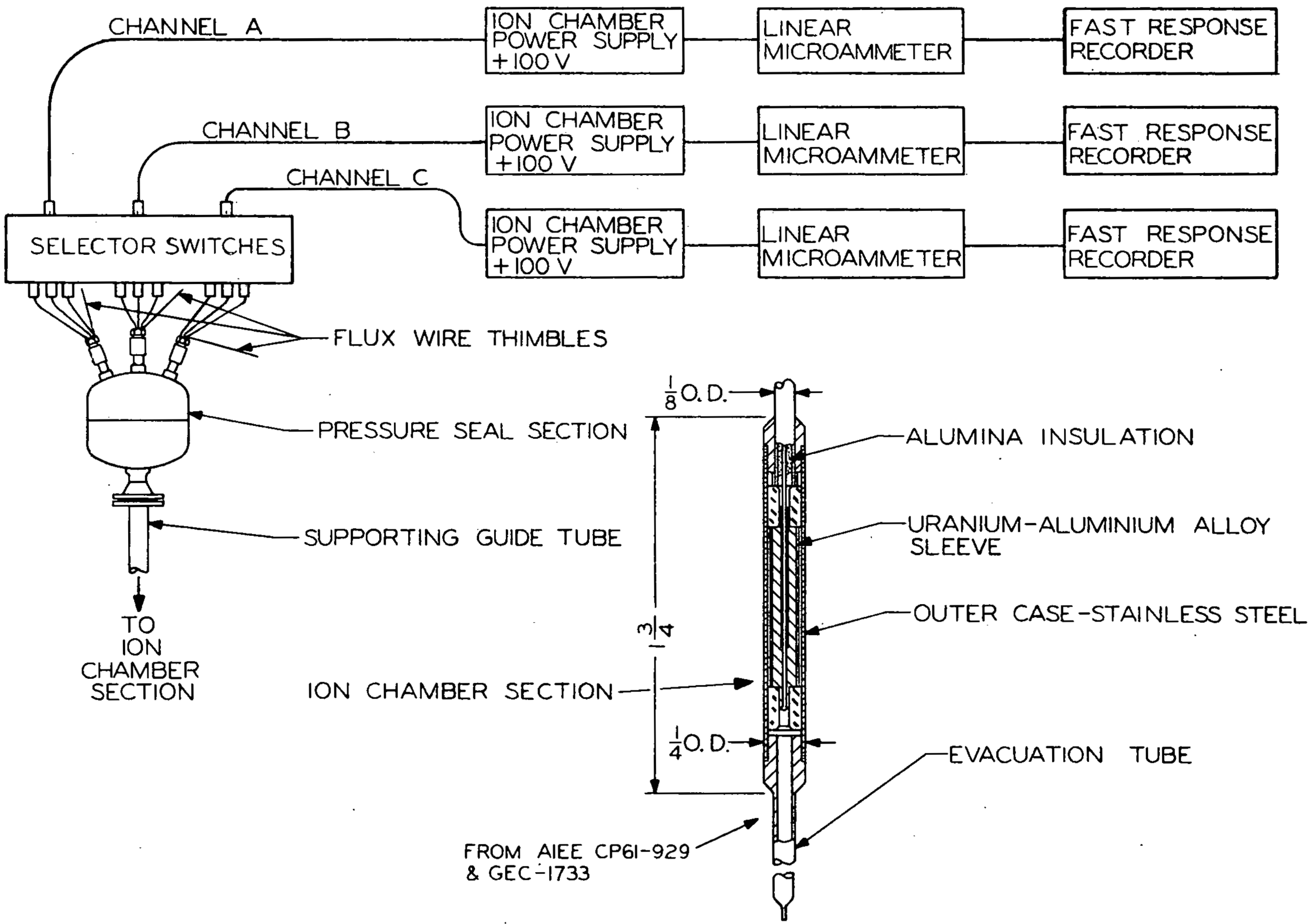
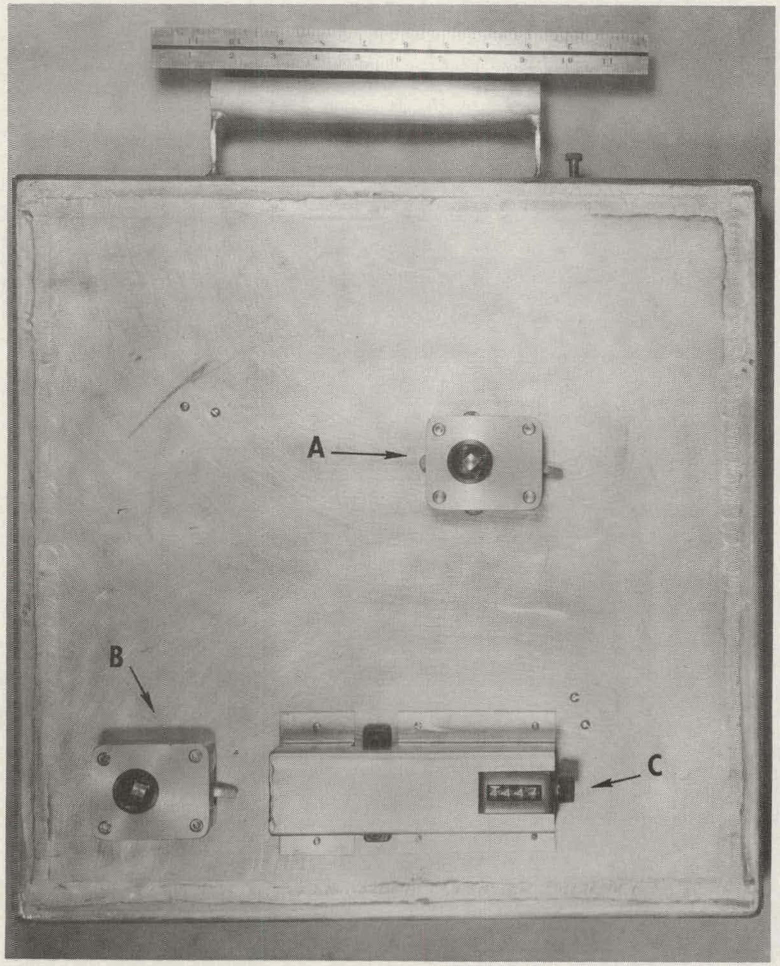


Fig. 1.2...In-Core Ion Chamber Instrumentation Block Diagram and Ion Chamber (43-024-933)

CALIBRATION WIRE

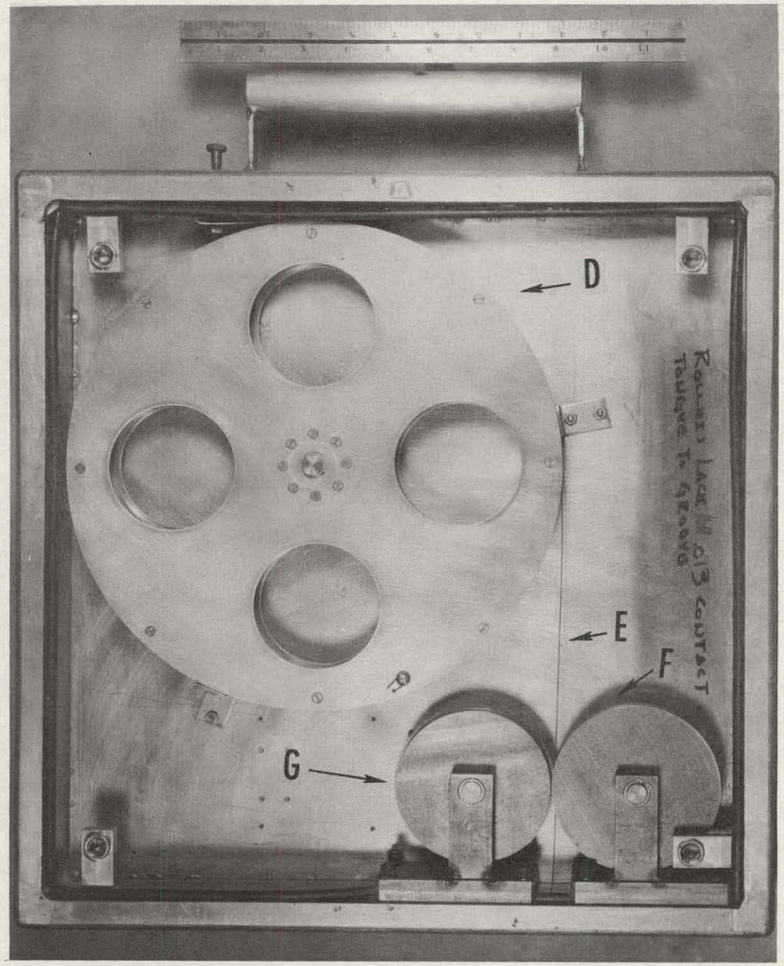
Figure 1.3 shows one of the three flux wire drive devices. These mechanisms will be manually operated to insert and withdraw the Cu-Ti calibration wire into the three flux wire thimbles. (One thimble is built into each ion-chamber bundle). Studies have shown that each of the Cu-Ti wires will be irradiated for about 20 minutes at full power. A decay period of about 12 hours will then be necessary in order to allow the Mn^{56} isotopes to decay. The backscatter of the Mn^{56} (.84) Mev γ interferes with the Cu^{64} .51 Mev γ which is the disintegration product that will be counted. Figure 1.4 shows the activity history of a section of a 6-inch piece of Cu-Ti wire that was irradiated in CP-5 at Argonne National Laboratory and that was subsequently counted in the Wire Counting Facility. This Wire Counting Facility is the same one that will be used at the reactor site to count the flux calibration wire. Thus, good estimates are now on hand for counter geometry effects and counting techniques. The total number of counts per inch of wire (integral count method) will be used to indicate wire activity.

Figure 1.5 is a block diagram of the Wire Counting Facility. This equipment, from the Allis-Chalmers Critical Facility, will automatically decay-correct the irradiated wire sample and with the drive motor, automatically drive the wire past the scintillation counter.



Exterior - Left Side (48D-0-5

(A) WITHDRAWAL RATCHET
(B) INSERTION RATCHET
(C) DISTANCE COUNTER



Interior - Right Side (48D-0-6)

(D) TAKEUP REEL (E) WIRE
(F) WIRE-DRIVE WHEEL (G) IDLER WHEEL & DISTANCE COUNTER DRIVE

Fig. 1.3...Manual Flux Wire Drives

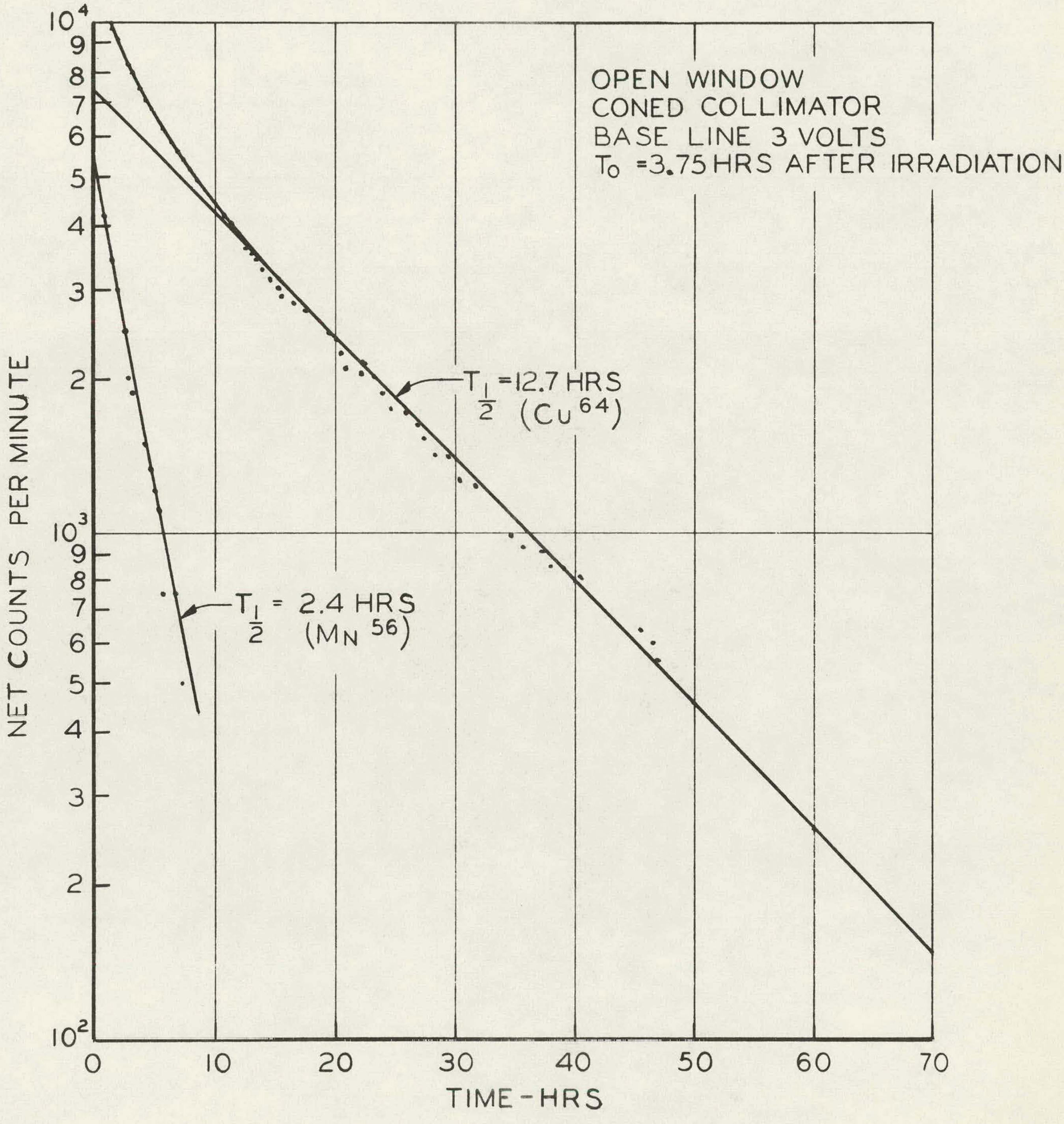
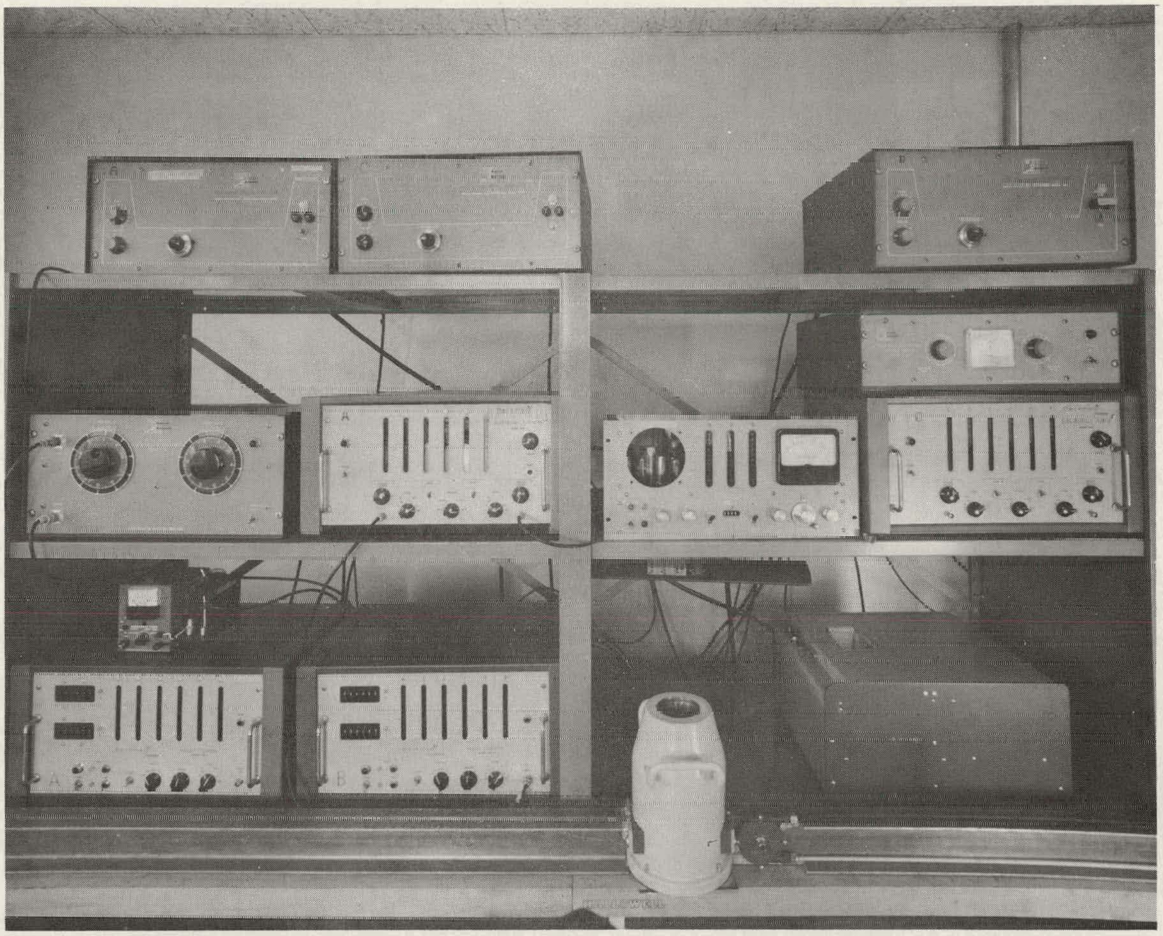


Fig. 1.4...Gross Decay of Activated Cu-Ti Wire Based at 30 kev with Coned Collimator (43-025-697)



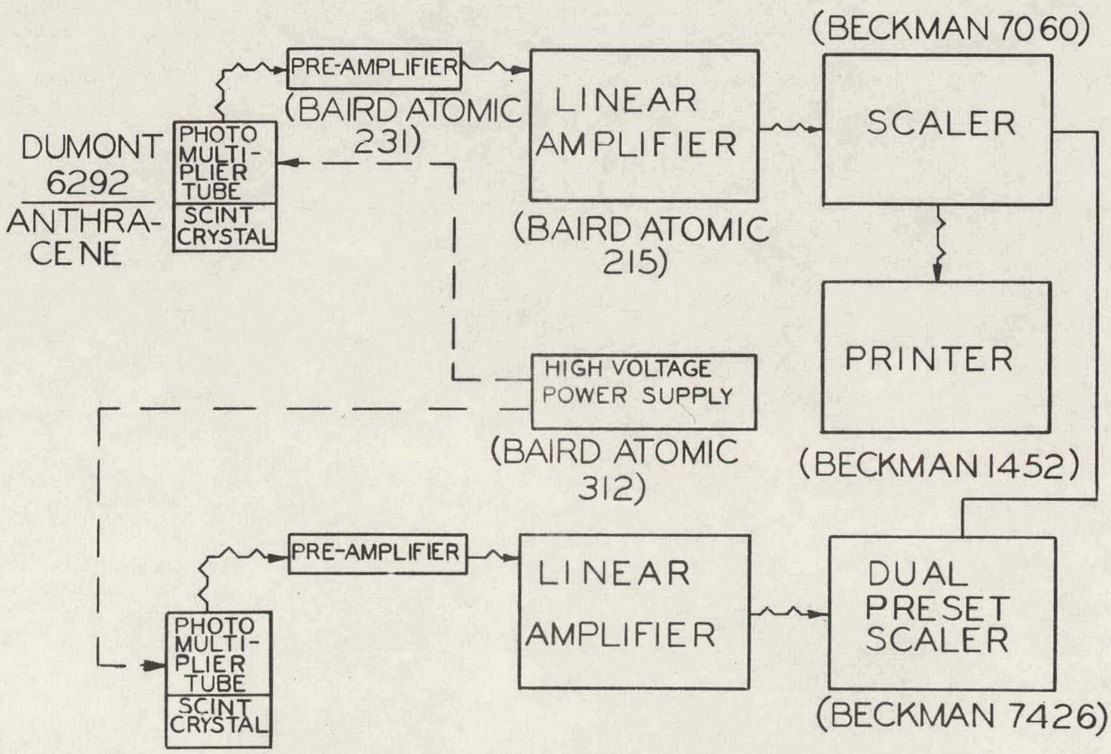


Fig. 1.5...Wire Counting Facility for In-Core Ion Chamber Calibration Wire (48D-0-1...43-025-694)

INSTRUMENTATION SYSTEM

The instrumentation system (shown in block diagram form in Figure 1.2) is shown in photo form in Figure 1.6. The three power supplies (battery system) will supply the 100 V for the ion chambers. The meters "A" will indicate the chamber voltage. The switches "B" will select the voltage desired for the chamber selected. (Note that three of the nine ion chambers can be simultaneously read and recorded). A battery charging system and simulated load are provided for battery protection. The micro-microammeter "C" (Keithley #414) will be used to convert the ion chamber current to a readable voltage. Although these are multi-range instruments capable of reading current from 10^{-11} to 10^{-3} amps, the low sensitivity of the ion chambers themselves will limit usefulness of this system to powers above about 7 MW(thermal) on Pathfinder. The switches "D" select the three of the 9 chambers that are to be recorded. In order to make a reasonable switching arrangement, switches were arranged so that 72 out of a possible 84 combinations of three could be selected. These are the most probable combinations of three.

The switches "D" operate a series of coaxial switches specially constructed so as to minimize noise and stray pickup in the switch box. Figure 1.7 shows a view of the coaxial switches and also a view of the battery power supply.

Figure 1.8 is an elementary block diagram of the in-core instrumentation cabinet.

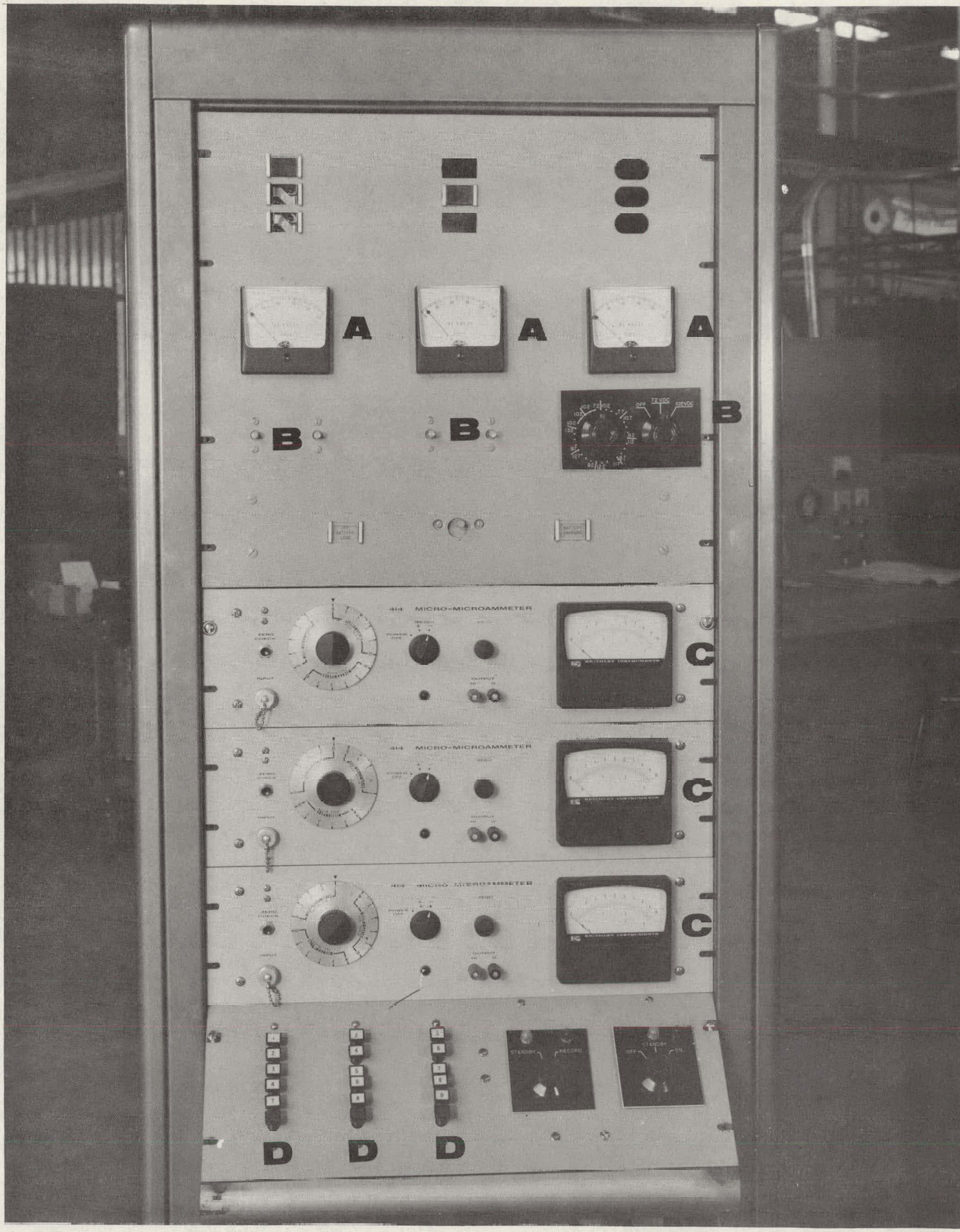
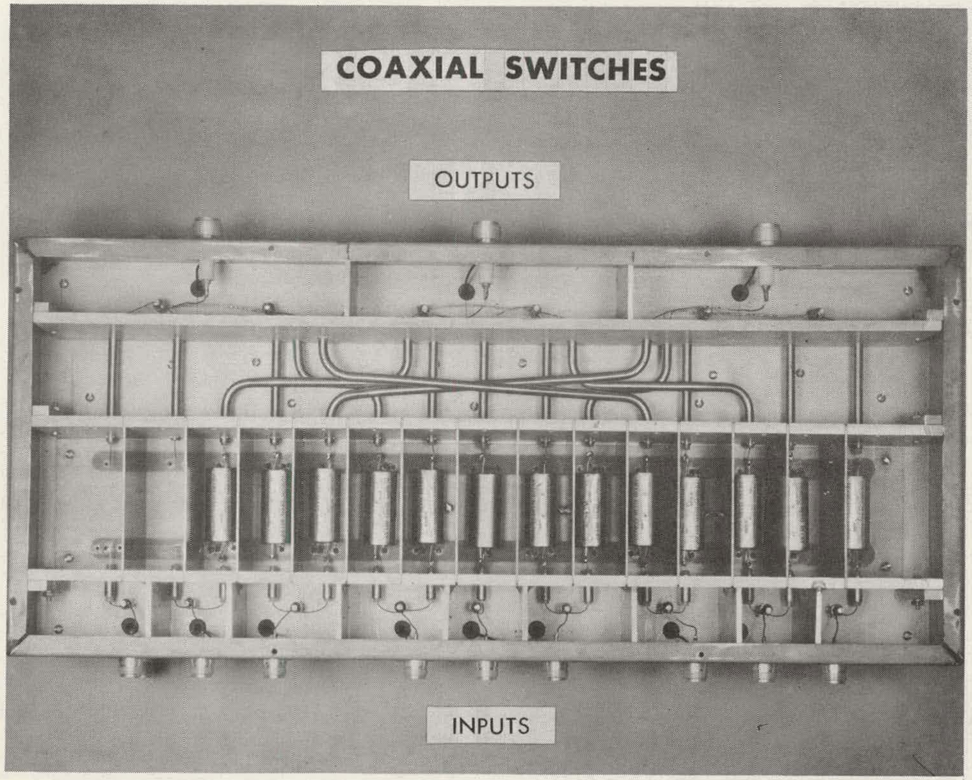


Fig. 1.6...In-Core Ion Chamber Instrumentation System (46D-0-4)



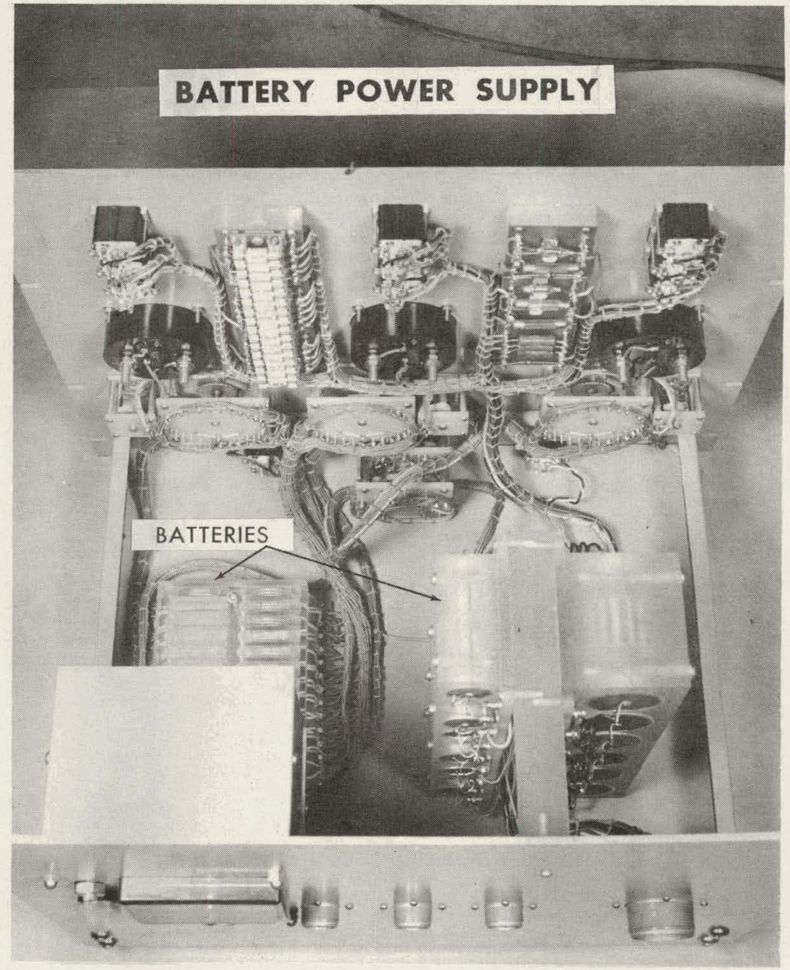


Fig. 1.7...Instrumentation System - Coaxial Switches and Battery Power Supply (48D-0-7...48D-0-10)

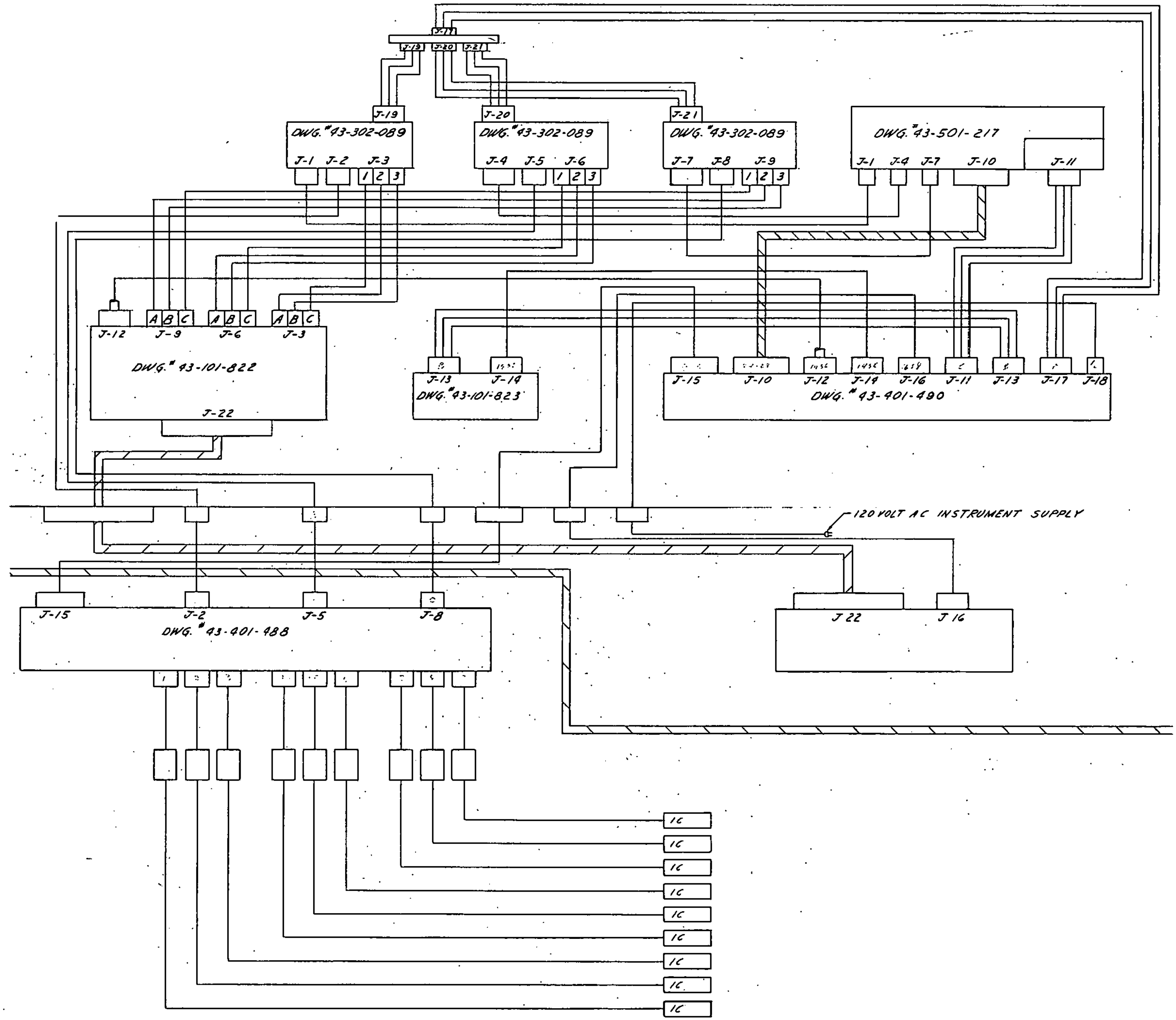


Fig. 1.8...In-Core Instrument Control and Monitor System (43-302-090)

OPERATION AND USE

During Phase I of the startup, a piece of the Cu-Ti wire will be irradiated in a known standard flux and its activity compared to the activity of wire irradiated in Pathfinder. (ACNP-6112, "Program and Organization Preoperational and Nuclear Testing," provides details of startup experiments). The flux in superheater and boiler fuel cells adjacent to the ion chamber locations will be calculated. Thus a correlation will be obtained between adjacent fuel cell fluxes and known wire readings. Corresponding superheater fuel surface temperatures will also be calculated in fuel cells adjacent to ion chamber locations. As a result, the calibration wire readings will be compared to calculated fluxes for future extrapolations (at power) of boiler heat fluxes and superheater fuel surface temperatures.

In Phase III, the above extrapolations will be carried out so that the ion chambers can be used to detect dynamic and static flux tilts in both the boiler and superheater, as well as local flux peaking. The calibration wire will, of course, be also used to calibrate the ion chambers as a function of reactor operating history.

During oscillation tests above 20 MW(thermal), phase differences of ion chambers signals in various directions will be observed; this may or may not give stability information, depending on the reliability of the data.

Throughout the power operation (above 10 MW) the temporary in-core chambers will be compared to the permanent out-of-core ion chambers to judge the usefulness of the in-core system as an aid to the reactor operator.

3.0 TESTING AND INSPECTION OF CORE COMPONENTS ...POST CONSTRUCTION R&D

The purpose of inspecting Pathfinder core components is to establish operation effects on mechanical, metallurgical, and chemical characteristics of these components.

3.0.1 GENERAL TEST METHOD

Selected core components will be removed from the core for underwater inspection in the storage pool or will be inspected in place in the core. If any defects or effects are encountered which are felt to be of sufficient interest, the component will be removed and shipped to a radiometallurgical facility for destructive examination. All such work will be carefully recorded and reported.

3.0.2 TEST PROCEDURE

3.0.2.1 Pre-Irradiation Measurements

During fabrication of the core components, appropriate measurements and other data will be recorded in the vendors' facility.

3.0.2.2 In the Case of Malfunctions

If at any time during Phases I, II and III operation of the reactor there appears to be a malfunction of a core component which reduces the operating performance of the reactor, the reactor will be shut down and the suspected component removed and subjected to a visual inspection.

3.0.3 INSPECTION PROCEDURES

If there are no apparent malfunctions, the inspection procedures summarized below will be incorporated.

3.0.3.1 Pathfinder Core I RML Post-Irradiation Inspection Program

This inspection program will be performed in the hot lab of Battelle Memorial Institute at Columbus, Ohio. The objectives of the inspection are to determine (1) core performance, (2) corrosion-crud analysis, (3) mechanical behavior and (4) material performance. Separate detailed inspection programs have been defined for the boiler and superheater core. The fuel elements will receive a cursory inspection under water prior to shipping to BMI.

Two boiler elements will be removed from the core at/or near end of life and shipped to the hot lab. The elements will be disassembled sufficiently to remove selected fuel pins for gamma scanning. Following gamma scanning, fuel samples will be extracted for burnup analysis. The experimental data will be used to determine the integrated power distribution within the fuel element. During disassembly, the elements will be given a visual examination. No other detailed examinations are planned unless a major defect is observed.

Three superheater fuel elements will be removed from the core at mid-life and end-of-life and shipped to BMI for post-irradiation inspection. The elements will be disassembled for gamma scanning and burnup analysis for determining integrated power distribution. The elements will receive a very thorough visual examination. At least one element will receive a detailed examination for crud and corrosion effects. The remaining elements will be examined to measure their mechanical behavior and to observe material performance.

3.0.3.2 Pathfinder Underwater Post-Irradiation Inspection Program

This program includes remote inspection, using an underwater periscope, of fuel elements and control rods, both boiler and superheater, and boiler boxes and superheater inner insulating tubes. Both direct visual and photographic techniques will be employed. The examination will help to determine the effects of reactor operation on the dimensional integrity and corrosion resistance of the components. It will also be used to "screen" the reactor components to help select those to be shipped out for the hot lab inspection.

3.0.4 CONTROL ROD INSPECTIONS

In order to more closely investigate the core performance of the control rods, the maximum-burnup boiler and superheater control rod will be pulled out and inspected at each refueling. The surfaces of the control rods will be examined for surface cracks and other discontinuities. The superheater rod will be checked for excessive swelling.

If any defects which are judged to adversely affect reactor operation are discovered, all the rods will be inspected in the same manner.

4.0 MECHANICAL TESTS...POST CONSTRUCTION R&D

The objectives of this project are to irradiate reactor vessel and steam line material samples and periodically examine them for embrittlement. Additional objectives are to determine the pre-startup flow distribution into the boiler core, to determine the two-phase interface in the operating reactor, and to determine the moisture carryover through the steam dryer into the superheater.

4.0.1 MATERIAL SURVEILLANCE AND INSPECTION

4.0.1.1 Reactor Vessel Material Irradiation Samples

There are provisions in the Pathfinder reactor vessel for 30 groups of material samples for material surveillance and other programs. Two sample holders are attached to each of 15 steam separator groups directly opposite the core. Fifteen (15) sample holders are located near the vessel wall and the remaining 15 are located near the vessel core for accelerated tests.

Each sample group may contain one or two sample capsules. Each stainless steel capsule, containing three Charpy specimens, is evacuated and pressurized with helium. Tight fitting aluminum spacers are used to improve heat transfer. Samples of the vessel material will be periodically removed from the reactor and tested throughout the lifetime of the reactor to detect the change in physical properties. Reference physical properties have been measured and were as follows: Tensile strength, 78,000 psi; yield point, 49,500 psi; elongation, 32%; R "A" hardness, 48.5-50. Transition temperature ductility curve is shown on Fig. 4.1. The frequency of such tests will be determined after the first samples, including the accelerated test samples nearer the

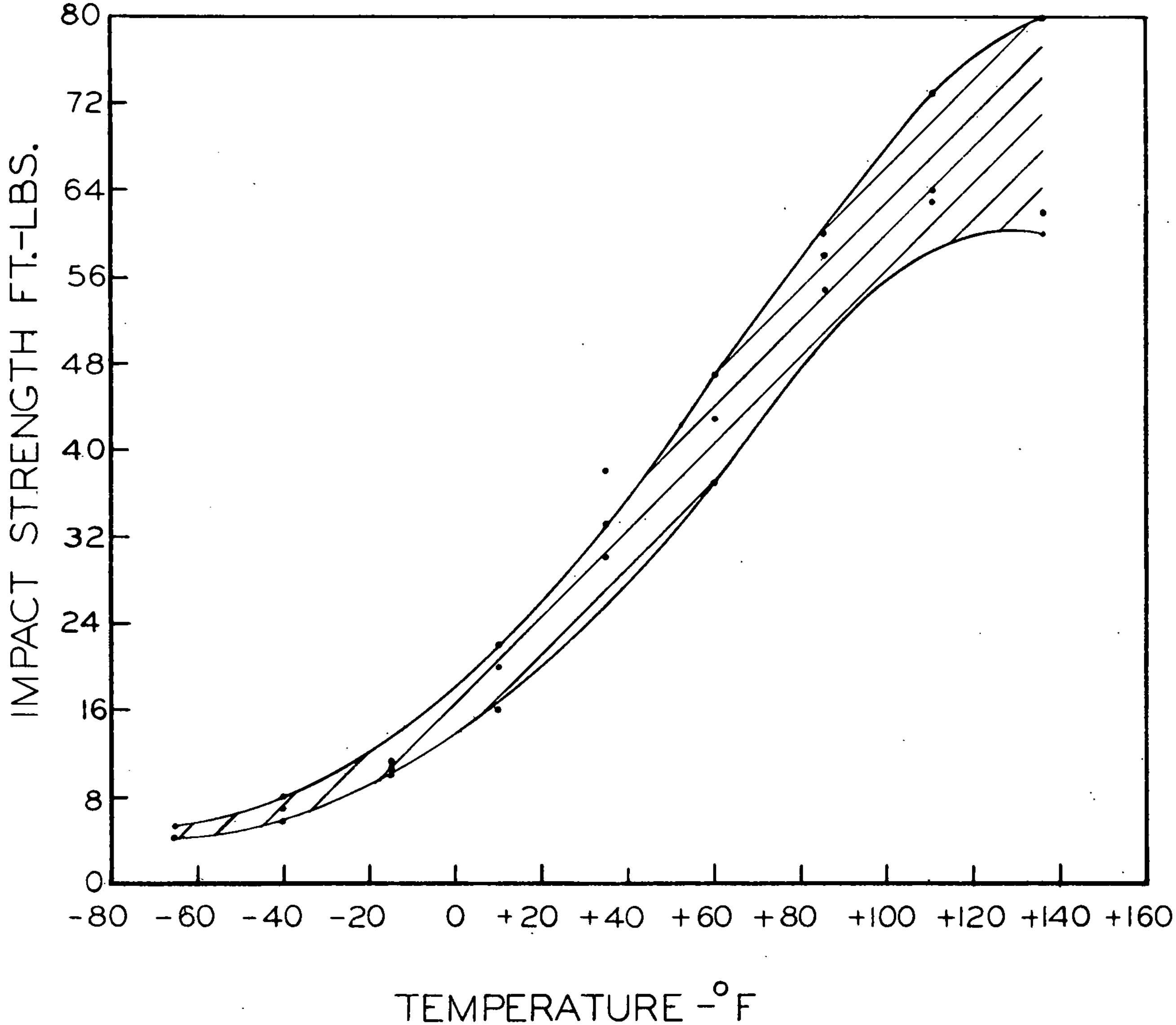


Fig. 4.1...Pathfinder Reactor Vessel ASTM A212B Steel Unirradiated Impact Strength (43-025-386)

core, have been removed and tested. Fifteen test capsules have been prepared and shipped to the site. Three contain eight (8) Charpy V-notch test specimens and six (6) tensile specimens each. Two (2) of these tensile specimens are stressed. The other twelve capsules contain eight (8) Charpy V-notch and two unstressed tensile specimens each. Figures 4.2 and 4.3 show test capsules for stressed and unstressed tensile specimens respectively.

4.0.1.2 Steam Line Irradiation Samples

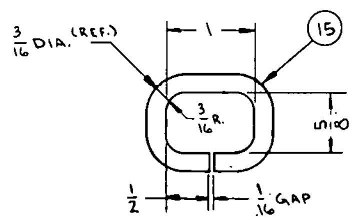
For steam line only impact strength will be surveyed. Charpy V-notch irradiation samples have been prepared from a section of the steam line that was removed at the containment vessel expansion joint. A number of the samples have been broken to establish the reference transition temperature ductility curve shown in Fig. 4.4. Fifty-six samples will be encapsulated, eight per capsule, and will be inserted for irradiation in a sample holder under the steam line elbow. A handling tool for removal of these samples has been designed and built. These samples will be periodically removed and tested, based on calculated exposure. Fast neutron exposure of all the samples will be determined by chemical separation of Mn^{51} from the $Fe^{54} + n = Mn^{54} + p$ transformation. The threshold for this reaction has been variously reported from 3.1 Mev to 5 Mev. There is evidence that this method may be reasonably extrapolated to include neutron energies as low as 1 Mev.

NOTE #1:-
ELONGATE ITEM #6 .0037" TO INDUCE STRESS.

NOTE #2:-
DEPTH OF FUSION FOR HELIARC SEAL WELDS TO BE EQUAL TO WALL THICKNESS

NOTE #3:-
TO BE WELDED ONLY AFTER EVACUATION PER IT. 16

MK. 501 REQ.	ITEM	DESCRIPTION	MATERIAL	PART NUMBER DRAWING	MK.	WT.
	1	IRRADIATION SAMPLE CONTAINER		43-401-388	501	[illegible]
1	2	LIFTING PIECE		43-202-556	502	
4	3	SPACER		43-101-609	001	
4	4	PIN HOLDER		43-202-552	001	
4	5	ROUND TENSION TEST SPECIMEN		43-202-559	502	
2	6	THREADED ROUND TENSION TEST SPECIMEN		43-202-558	502	
4	7	SHIM		43-202-555	001	
8	8	CHARPY V-NOTCH TEST SPECIMEN		43-202-333	502	
4	9	1/4-28 UNF-2B HEAVY SEMIFINISHED HEX. NUT		Q0006804		
1	10	BOTTOM CAP		43-202-557	502	
1	11	PICKUP RING		43-202-339	502	
~~1~~	~~12~~	~~1/8 WIRE x 20 LINKS/FT. x 89 LG. WELDED LINK CHAIN~~	~~STN. STL. 304~~	~~THIS~~	~~012~~	
1	13	CONTAINER		43-202-554	504	
.01#	14	WELD WIRE (ASTM A371-53T CL. ER 308)		THIS	014	
1	15	ATTACHMENT RING - 3/16 DIA. x 4 LG. BAR	ASTM A-276 TP-304	THIS	015	
YES	16	EVACUATING SPEC.		43-101-[illegible]	401	
YES	17	MARKING SPEC.		43-101-282	401	
YES	18	FABRICATION SPEC.		43-101-216	401	
YES	19	CLEANING SPEC.		43-201-661	402	
1	20	#2/0 (.135) x 6 1/3 LINKS/FT x 89 LG. DOUBLE LOOP WELDLESS CHAIN (BULL. #59 PAGE 20 S.G. TAYLOR CHAIN CO. PITTSBURGH PA)	STN STL 304	THIS	020	



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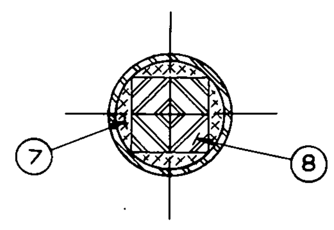
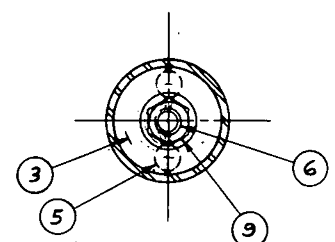
SCALE: 12:12



SECTION "A-A"



SECTION "B-B"

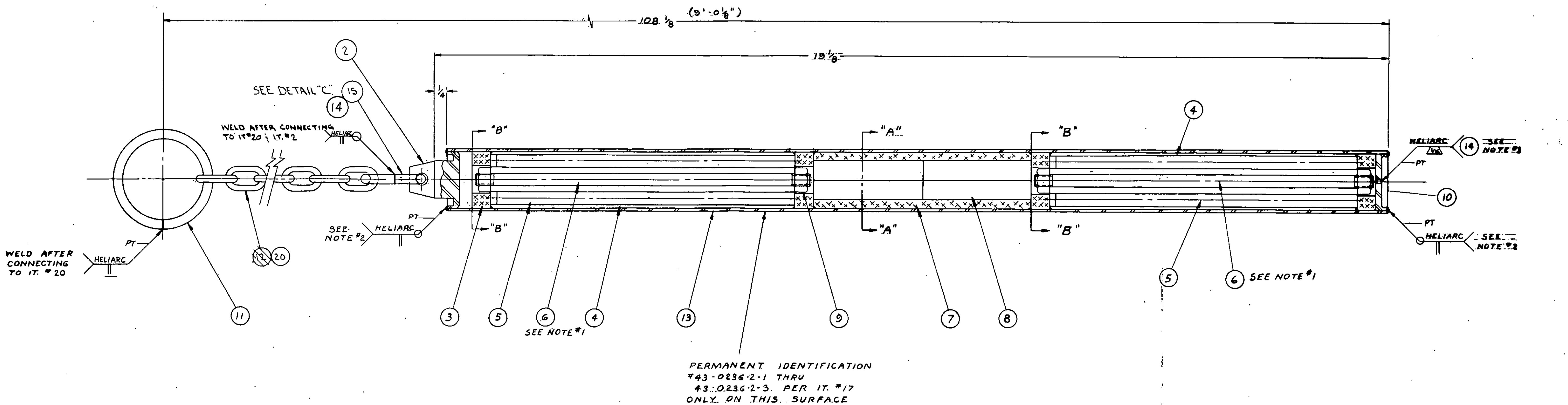


Fig. 4.2...Irradiation Sample Container for Stressed Tension Specimens (43-401-388)

MK. 501 REQ.	ITEM	DESCRIPTION	MATERIAL	PART NUMBER DRAWING	MK.	WT.
	1	IRRADIATION SAMPLE CONTAINER		43-401-389	501	3.5#A.
1	2	LIFTING PIECE		43-202-556	502	
2	3	PIN HOLDER		43-202-553	001	
2	4	ROUND TENSION TEST SPECIMEN		43-202-559	502	
4	5	SHIM		43-202-555	001	
8	6	CHARPY V-NOTCH TEST SPECIMEN		43-202-338	502	
1	7	BOTTOM CAP		43-202-557	502	
1	8	PICKUP RING		43-202-359	502	
~~1~~	~~9~~	~~1/8 WIRE × 20 LINKS/FT. × 97" LG. WELDED LINK CHAIN~~	~~STN. STL 304~~	~~THIS~~	~~009~~	
1	10	CONTAINER		43-202-554	502	
.01#	11	WELD WIRE (ASTM A 371-53T CL. ER 308)		THIS	011	
1	12	ATTACHMENT RING - 3/16 DIA. × 4 LG. BAR	STN. STL. 304	THIS	012	
YES	13	EVACUATING SPEC		43-101-501	401	
YES	14	MARKING SPEC.		43-101-282	401	
YES	15	FABRICATION SPEC.		43-101-216	401	
YES	16	CLEANING SPEC.		43-201-661	402	
1	17	2/0 (.135) × 6 1/3 LINKS/FT × 97 LG. (BULL. 154 PAGE 30) DOUBLE LOOP WELDLESS CHAIN (S.G. TAYLOR CHAIN CO. PITTSBURGH PA.)	STN. STL. 304	THIS	017	

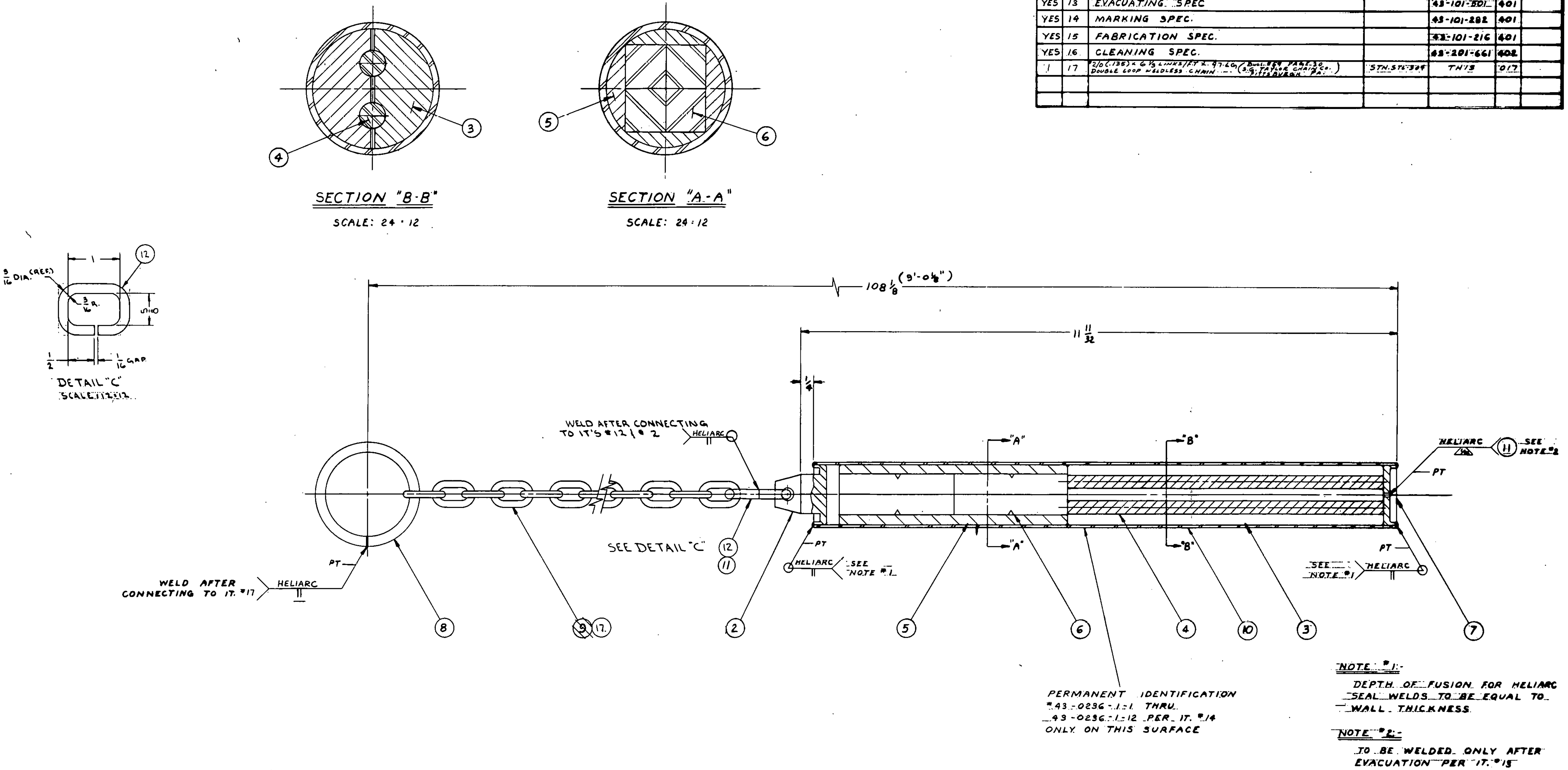


Fig. 4.3...Irradiation Sample Container for Unstressed Tension Specimens (43-401-389)

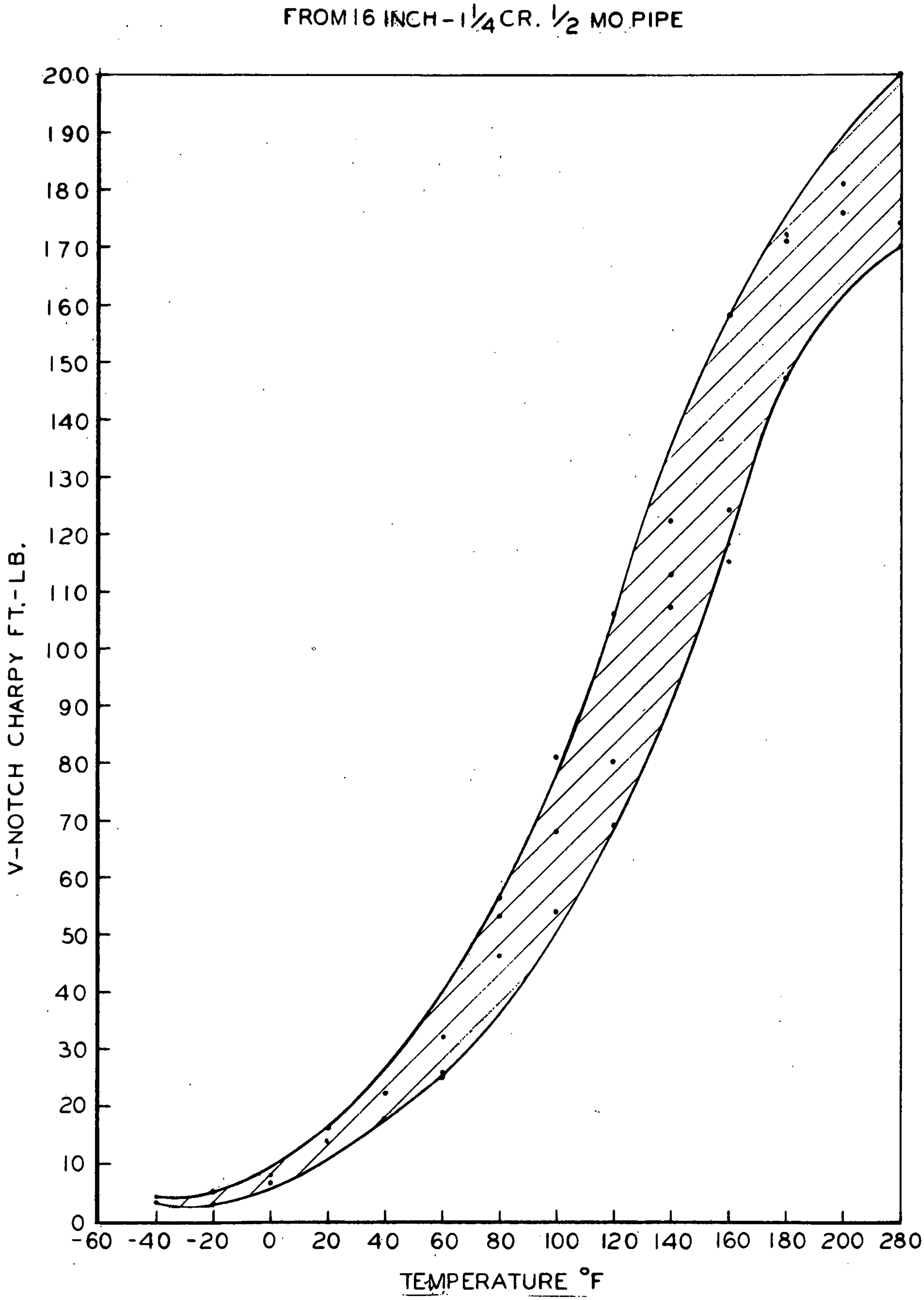


Fig. 4.4...Transition Temperature Ductility Curve for Pathfinder Steam Line (43-025-662)

4.0.2 INSTRUMENTATION

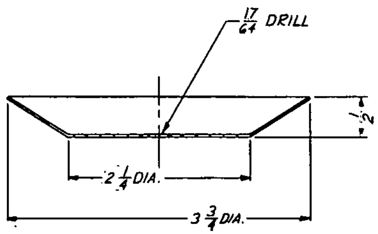
Instrumentation for measuring the flow distribution in the boiler core has been designed and shipped to the site. It consists of an impact tube - manometer setup. The impact tube will be placed over representative boiler core orifices and the relative velocity head will be measured for various recirculation pump combinations. This will be compared with the results of a mockup of the recirculation loop reported in ACNP-5920 in which air was used to simulate water.

4.0.3 TWO-PHASE INTERFACE

The two-phase interface will be determined by measuring the flow through three detectors located in the reactor vessel and positioned at 1 ft 3 in., 2 ft 9 in., and 3 ft 3 in. of indicated reactor water level. The detector consists of a tube with the end plugged and with a 0.040 in. hole drilled in its wall (Fig. 4.5).

In order to lessen the effect of turbulence at the two-phase interface a filter and a baffle surround the orifice. From the orifice the flow passes out of the reactor via a tube, is condensed and is measured on a rotometer type flow meter. The flow rate through the orifice is a function of its position above the two-phase interface. This flow rate has been experimentally determined in a series of tests in which steam was bubbled through water at various pressures. The steam rates simulated those of the Pathfinder reactor. These results will be used to correlate the steam-water interface in the Pathfinder and are shown in Fig. 4.6.

MK. 501 REQ	ITEM	DESCRIPTION	MATERIAL	PART NUMBER DRAWING	MK	WT.
/	1	LIQUID LEVEL DETECTOR		43-301-974	501	
2	2	DISH-M/F #22 GA(.031) × 5 × 5 LG SHEET	ASTM-A240 TYPE 304	THIS	002	
1	3	TUBE-M/F 1/4 O.D. × 18 GA.(.049) WALL × 12 LG. TUBE	ASTM-A269 TYPE 304	THIS	003	
1	4	PLUG-M/F 5/32 O.D. × 1/8 LG. BAR	ASTM-A276 TYPE 304	THIS	004	
1	5	WELDED WELL SCREEN		43-101-627	501	
YES	6	CLEANING SPEC.		43-201-661	402	
YES	7	WELD PROCEDURE SPEC.		43-101-604	401	
YES	8	MARKING SPEC.		43-101-282	401	
1	9	SOCKET WELD UNION-1/4 TUBE (SWAGELOCK #400-6-4SW-316)	STN. STL. TYPE-316	THIS	005	



DETAIL OF IT. #2

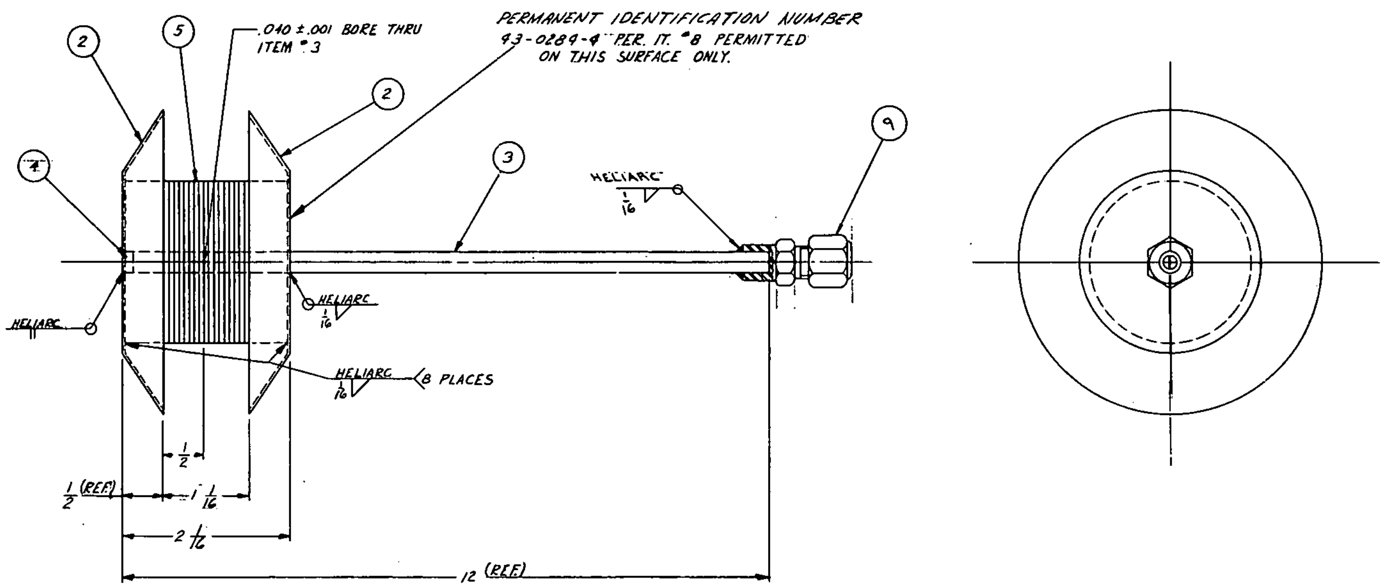


Fig. 4.5...Liquid Level Detector (43-301-974)

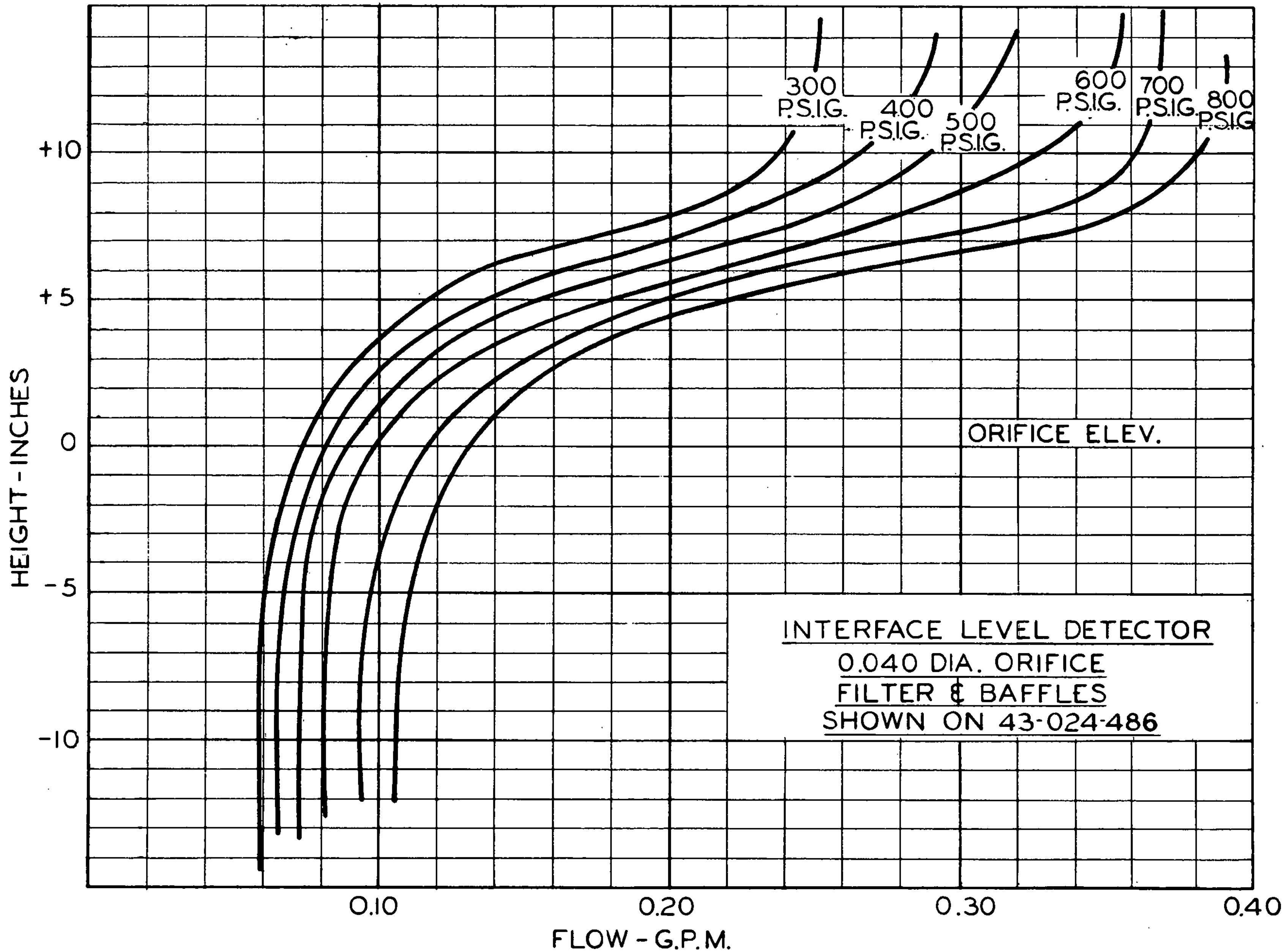


Fig. 4.6...Flow Rate vs Interface Location (43-025-387)

4.0.4 STEAM DRYER EFFICIENCY

The steam dryer efficiency will be determined by measuring the moisture content of the steam before and after it passes through the steam dryer. The quality of steam from sampling point ahead of the steam dryer is determined in a throttling calorimeter. The steam from sampling point after the steam dryer, passes through a superheating calorimeter. Here the steam is electrically heated to a few degrees of superheat and the temperature measured. The power input to the steam is measured with an accurate wattmeter. After leaving the calorimeter, the steam is condensed and its flow rate is measured on a rotometer-type flowmeter. From these results, its quality is determined.

4.0.5 LIQUID LEVEL PANEL BOARD

Instrumentation for two-phase interface and steam dryer efficiency determination has been assembled onto a panel board (Fig. 4.7) and has been shipped to the Pathfinder site.

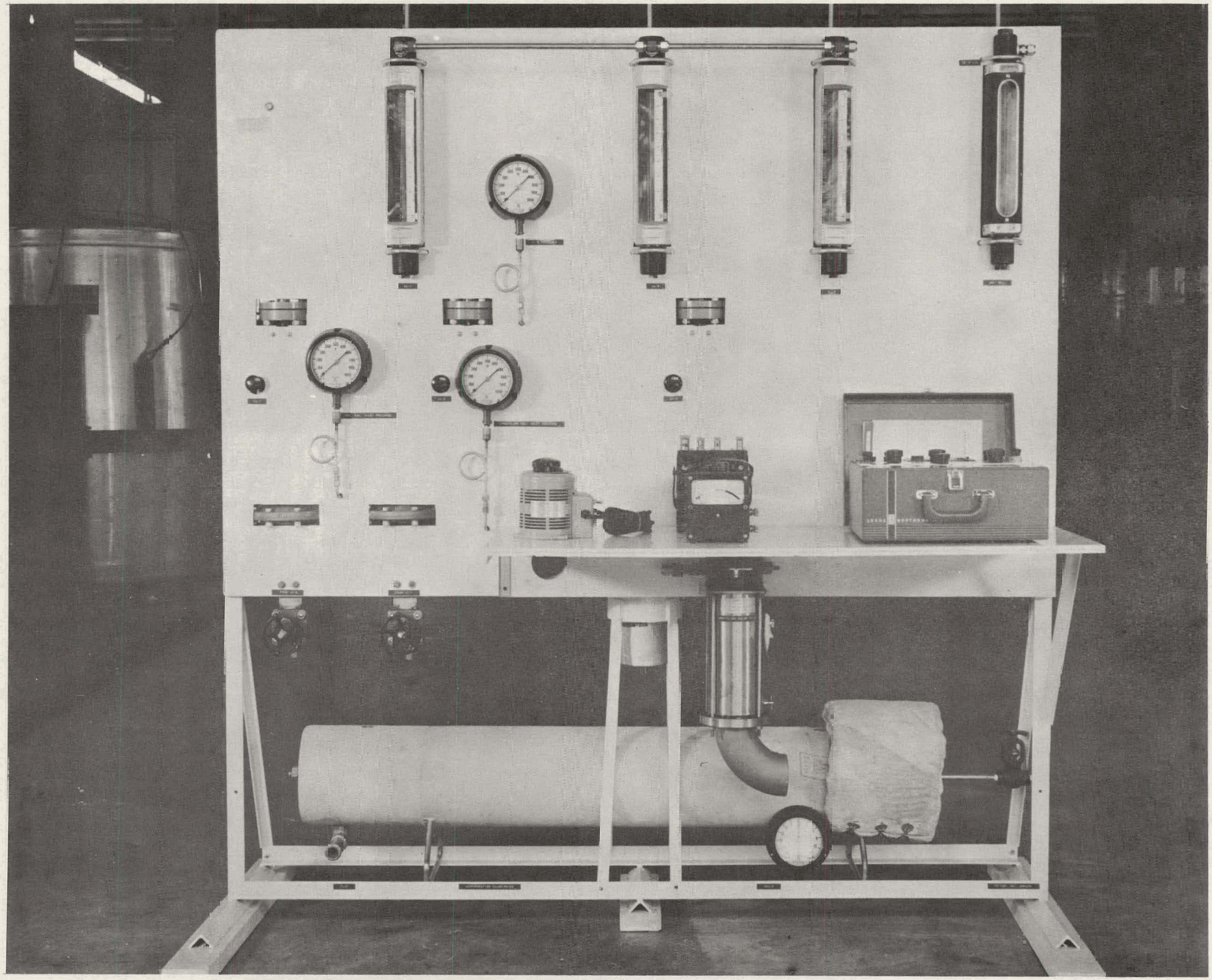


Fig. 4.7...Liquic Level Panel Board (39D-0-1)

6.0 PHYSICS, POST CONSTRUCTION R&D

An initial startup program for Pathfinder has been specified. Test procedure specification and pre-testing analysis are currently in progress. In subsequent quarters, the results of the pre-testing analysis, experiments, and final analysis will be covered in Pathfinder Quarterly Reports. In this current report, the principal physics tests, their purpose, and the experimental procedure are briefly described.

6.0.1 INTRODUCTION

The initial nuclear startup program for the Pathfinder Atomic Power Plant consists of a series of tests designed to demonstrate the physics performance of the reactor and the validity of the analytical model used in the reactor physics design. These tests are sequenced in their order of increasing reactor power level i.e., all tests at a given power level are performed before the power is raised to the next higher level. Results from the design analysis and critical facility experiments are utilized and compared with measured results from this program. Results from each test are used to predict the performance and safeguards evaluation of subsequent tests.

6.0.2 INITIAL CORE LOADING

To provide a basis for verification of the analytical methods used to compute criticality of the Pathfinder reactor and for management of fuel loading in subsequent operations, initial criticality is achieved in steps for core configurations of increasing complexity from the minimum critical boiler fuel mass to the full integral superheat core.

6.0.2.1 Boiler Slab Core Loading and Criticality

The number and configuration of the boiler elements for this minimum mass criticality test are to be predicted by calculations and extrapolation of Pathfinder critical facility slab experiments.

Three startup chambers are located in the vessel outside the boiler fuel region. A 6-curie Pu-Be neutron source is located at the periphery of the superheater region. Ten 2.2 w/o U-235 enriched boiler fuel elements are loaded into the dry core to form a slab array. The neutron population is monitored during the loading. A control rod worth about 4 per cent $\Delta k/k$ is cocked. The neutron multiplication is monitored as water is added to the vessel. At specified levels, water addition is stopped and count rates are taken. The maximum rate of flow shall not exceed that which corresponds to a maximum reactivity addition rate of about two cents per second for multiplication (k) values greater than 0.8.

The calculated shutdown margin of this minimum critical mass assembly with all control rods full in is 14 per cent $\Delta k/k$. The slab core loading and water fill are performed as if criticality is expected at any point even though the fully moderated assembly is expected to be at least 10 per cent $\Delta k/k$ sub-critical with a rod cocked (4 per cent $\Delta k/k$). At any point the water level can be reduced. Criticality is not to be achieved at any time during the fuel loading or water fill. If criticality is predicted at any time, the core region is to be drained and fuel removed to reduce reactivity.

Criticality is approached by the withdrawal of boiler control rods. If criticality is not achieved, the core is drained and a single 2.2 w/o boiler element added to the slab. The core is again water-filled and criticality is approached by control rod withdrawal. These steps are repeated adding a single 2.2 w/o boiler element at each step until criticality is achieved. After initial criticality is achieved, the core is drained, and the test is repeated adding an additional 2.2 w/o boiler fuel element to determine the incremental increase in core reactivity associated with a single boiler element.

Initial criticality is not anticipated with the first slab loading since the number of fuel elements in the loading is to be one less than the number of elements predicted to form a critical mass, giving due consideration to uncertainties. However, if initial criticality is achieved on the first slab loading, the core is to be drained, one element removed and the test repeated until the minimum critical mass is reached.

6.0.2.2 <u>Boiler Full Core Loading Criticality</u>

With the core drained, boron-stainless steel poison shims worth approximately 4 per cent $\Delta k/k$ are inserted between predetermined boiler fuel boxes. The 2.2 w/o boiler elements are added to complete the boiler core (96 2.2 elements). The neutron population is monitored during this process.

Control rods worth about 4 per cent $\Delta k/k$ are cocked. The neutron multiplication is monitored as water is added to the vessel. At specified levels, water addition is stopped and count rates are taken.

The calculated shutdown margin of this assembly with all rods and 4 per cent $\Delta k/k$ in poison shims inserted is approximately 10 per cent $\Delta k/k$. Criticality is not to be achieved at any time during fuel addition or water fill. If criticality is predicted at any time, the core region is to be drained and additional poison shims added to reduce reactivity.

Criticality is to be achieved by the withdrawal of control rods. After the reactor is critical at a low power, a rod drop measurement is performed to estimate the shutdown margin. The poison shims are removed in steps, and shutdown data are recorded for each step.

6.0.3 PHASE I 200 KW (TH) OR LESS

Phase I experiments are performed at low power and ambient conditions to establish the reference core. Critical control rod configurations, core power distributions, and reactivity coefficients are measured for the reference core. Control rods are calibrated during the performance of those tests which require calibrated rod data for test interpretation. These experimental results are to be compared with calculations. These measurements and comparisons are also to provide verification of the shutdown reactivity margin and thermal margin and the validity of the analytical model used for the design calculations of Pathfinder.

6.0.3.1 Superheater Fuel Loading and Full Core Criticality

All poison shims (worth approximately 6½ per cent $\Delta k/k$) are loaded into the dry core. The superheater fuel is loaded, and the neutron population is monitored during this loading. Control rods worth about 4 per cent $\Delta k/k$ are cocked. The neutron multiplication is monitored as water is added to the vessel. At specified levels, water addition is stopped and count rates are taken.

The superheater steam passages remain voided (most reactive configuration) during water addition. The calculated shutdown margin of this assembly with all rods and poison shims inserted is approximately 10 per cent $\Delta k/k$. The superheater fuel loading and water fill are performed as though criticality were expected at any point, even though the moderated assembly is expected to be at least 6 per cent $\Delta k/k$ sub-critical with the cocked rod pattern (4 per cent $\Delta k/k$). At any point the water level can be reduced.

Criticality is to be achieved by the withdrawal of boiler control rods. After the reactor is critical at a low power, a rod drop measurement is performed.

6.0.3.2 Establishment of the Reference Core

The reference core is defined as that full core which is just sub-critical with the most reactive control rod withdrawn and the superheater voided.

Poison shims are removed incrementally until this criterion is met, thereby establishing the reference core. If the most-reactive-rod-withdrawn criterion is met with substantial shutdown margin, positive shim boiler fuel elements (3.2 w/o U-235 enrichment) are available to increase the core excess reactivity for operational purposes. During the establishment of the reference core, all control rods in shutdown data are also recorded as a function of poison shim removal.

6.0.3.3 Reference Core Cold Flooding Coefficient

Because of the superheater central location, the effect on core reactivity of flooding the superheater is of interest. With the reactor shut down, the superheater steam passages are flooded. The reactor is brought to criticality and the reactivity defect associated with flooding the steam passages is evaluated by means of a calibrated boiler control rod.

6.0.3.4 Core Flux Mapping

At each of approximately seven different control rod configurations, U-Al flux wires are loaded into the superheater and boiler. The concentration of boron added to the moderator is varied to provide control rod configurations typical of those to be encountered during operation from reactor startup to the control-rods-full-out condition. The data are analyzed for superheater-boiler power sharing, axial power shapes, symmetry of core radial power distribution, boiler and superheater radial power peaks, and nuclear instrumentation calibration.

6.0.3.5 Insertion of Reactor Source and Refueling Test

The 6-curie Pu-Be source is used initially for health physics considerations during the dry loading and for easier testing. Preparatory to further testing, the reactor source (10^{10} n/sec) is loaded in the core, and the Pu-Be source is removed. The reactor vessel is completely filled, including the superheater steam passages, during this operation. Reference core criticality is repeated to correlate the regular instrumentation indication to known conditions.

For refueling purposes, the maximum reactivity increase associated with the insertion of a boiler fuel element into a water-filled box is determined. Several boiler box locations with all rods in are to be investigated. The worth of both 2.2 w/o and 3.2 w/o boiler elements is to be determined.

6.0.3.6 Cold Core Pressurization

The system is gradually pressurized. A calibrated boiler control rod is used to determine the reactivity change resulting from the pressurization over 50 psi increments to 600 psi.

6.0.3.7 Temperature Coefficient

The temperature coefficient is measured from ambient to 440 F. With the superheater flooded, the reactor is pressurized to 600 psig to prevent boiling during this test. The reactor is taken critical and a slow heating rate is established with the startup heater. A single boiler control rod is withdrawn to yield a 30-second period - and then inserted. The temperature is increased by 10 F...the same boiler rod withdrawn to the same position...and the period measured. Criticality is re-established and a second boiler rod withdrawn to yield a 30-second period, and so on.

6.0.3.8 Hot Core Flooding Coefficient

The reactivity change associated in flooding the superheater steam passages with approximately 440 F water is determined by means of a calibrated control rod. This test is performed in the same manner as the cold core flooding coefficient.

6.0.4 FULL POWER OR LESS

Full power is approached in about five steps. At each power level the following physics tests are performed.

6.0.4.1 Radiation Testing

Radiation level data are taken at many positions throughout the plant. In addition to health physics considerations, the data are taken to verify the analytical model used for Pathfinder shielding calculations.

6.0.4.2 Xenon Reactivity

The reactivity associated with transient xenon is determined at several power levels by operation until near equilibrium poisoning is reached. Then the reactor power level is reduced substantially and the resultant reactivity charges followed with control rod movement until essentially all xenon has decayed. Rod calibrations are performed during the transient follow. The effects of samarium transients are included by appropriate analysis of the operating history.